ATTACHMENT B

TECHNICAL SPECIFICATION CHANGES FOR LASALLE UNIT 1

SUMMARY OF PROPOSED CHANGES FOR LASALLE UNIT 1 (NPF-11)

Section 3.4.2 Page 3/4 4-5	SRV safety valve function lift setting tolerances changed from +1%, -3% to ±3%; SRV as-left safety valve function lift setting tolerances specified to be ±1%

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of 17 of the below listed 18 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift setting*#; all installed valves shall be closed with OPERABLE position indication.

a. 4 safety/relief valves @ 1205 psig +1%, -3%
b. 4 safety/relief valves @ 1195 psig +1%, -3%
c. 4 safety/relief valves @ 1185 psig +1%, -3%
d. 4 safety/relief valves @ 1175 psig +1%, -3%
e. 2 safety/relief valves @ 1150 psig +1%, -3%

±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 110°F, close the stuck-open relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more of the above required safety/relief valve stem position indicators inoperable, restore the inoperable stem position indicators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.4.2.1 The safety/relief valve stem position indicators of each safety/relief valve shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 31 days, and a
b. CHANNEL CALIBRATION at least once per 18 months.**

4.4.2.2 The low-low set function shall be demonstrated not to interfere with the OPERABILITY of the safety relief valves or the ADS by performance of a CHANNEL CALIBRATION at least once per 18 months.

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*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

#Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling outage.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

Following testing, lift settings shall be within ±1%.

ATTACHMENT C

SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated the proposed Technical Specification Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of LaSalle County Station Unit 1 in accordance with the proposed amendment will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

   The probability of an accident previously evaluated will not increase as a result of this change, because the only changes are the tolerances for the SRV opening setpoints and the speed of the RCIC turbine and pump. Changing the maximum allowable opening setpoint for the SRVs does not cause any accident previously evaluated to occur, or degrade valve or system performance in any way so as to cause an accident to occur with an increased frequency. In addition, the increased speed of the RCIC turbine and pump are within the design limits of the system. RCIC operability and failure probabilities are not impacted by this change. This is supported by the Safety Analysis (Attachment A) and in Attachments E and H (GE and S&L Analyses, respectively).

   The consequences of an ASME Overpressurization Event are not significantly increased and do not exceed the previously accepted licensing criteria for this event. GE has calculated the revised peak vessel pressure for LaSalle Station to be 1341 psig, which is well below the 1375 psig criterion of the ASME Code for upset conditions, referenced in Section 5.2.2, Overpressurization Protection, of the Updated Final Safety Analysis Report (UFSAR), and NUREG-0519 (Safety Evaluation Report related to the operation of LaSalle County Station, Units 1 and 2, March 1981), and Section 15.2-4, Closure of Main Steam Isolation Valves (BWR) of NUREG-0800 (Standard Review Plan). The consequences of this event will continue to be verified on a cycle-specific basis, beginning with L1C8. These analysis results will be approved as part of the normal reload licensing 10CFR50.59 processes.

   GE has also performed an analysis of the limiting Anticipated Transient Without Scram (ATWS) event, which is the MSIV Closure Event. This analysis calculated the peak vessel pressure to be 1457 psig, which is well below the 1500 psig criterion of the ASME Code for emergency conditions.

Per NUREG-0519, listed above, Section 5.4.1, and Technical Specification 4.7.3.b, the RCIC pump is required to develop flow greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is supplied to the turbine at 1000 +20, -80 psig. Increasing the turbine and pump speed ensures these criteria will still be met and the consequences of an accident will not increase.

The conclusions given in Attachments A, E, and H with regards to containment dynamic loads, high pressure system performance, main steam piping loads, LOCA impact and MCPR impact also show that current accident and transient analyses are not impacted by this change beyond those reanalyzed by GE in Attachment E.

Therefore, there is not a significant increase in the consequences of an accident previously evaluated.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated because:

The only physical changes are to increase the allowable tolerances for SRV opening setpoints and to increase the RCIC pump and turbine speeds. These changes do not result in any changed component interactions. The SRVs and RCIC will still provide the functions for which they were designed. Since all of the other systems evaluated in Attachments A, E, and H will continue to function as intended, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3) Involve a significant reduction in the margin of safety because:

While the calculated peak vessel pressures for the ASME Overpressurization Event and the MSIVC ATWS Event are larger than that previously calculated without the proposed setpoint tolerance increases, the new peak pressures remain far below the respective licensing acceptance limits associated with these events. In addition, the actual L1C8 reload analysis of the ASME Overpressurization Event will be verified to be within the licensing acceptance limit for that event prior to Unit 1 Cycle 8 startup, as required in the normal reload 10CFR50.59 process. These licensing acceptance limits have been previously evaluated as providing a sufficient margin of safety. For other accidents and transients, the increased setpoint tolerances have a negligible, if any, effect on the results, so the margin of safety is preserved.

Guidance has been provided in "Final Procedures and Standards on No Significant Hazards Considerations," Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations. These proposed amendments most closely fit the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the applicable Standard Review Plan.

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92 (c), the proposed change does not constitute a significant hazards consideration.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonwealth Edison has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluents that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.

ATTACHMENT H

SARGENT AND LUNDY ANALYSIS

FOR

SRV DISCHARGE PIPING AND MAIN STEAM PIPING LOADS

DUE TO SRV SETPOINT TOLERANCE RELAXATION